NON-PUBLIC?: N
ACCESSION #: 9010260079
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Grand Gulf Nuclear Station PAGE: 1 OF 4

DOCKET NUMBER: 05000416

TITLE: Reactor Scram Due to Loss of BOP Busses
EVENT DATE: 09/16/90 LER #: 90-017-01 REPORT DATE: 10/11/90

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Riley Ruffin/Licensing Specialist TELEPHONE: (601) 437-2167

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

A Division I Load Shedding and Sequencing System malfunction caused a Balance of Plant (BOP) load shed on September 16, 1990. The loss of major plant equipment, which received power from the shedded BOP busses, resulted in a reactor scram, due to Main Turbine Control Valve fast closure. Subsequent to the scram, reactor water level decreased to -41.6 inches where an automatic , High Pressure Core Spray System actuation occurred. During restoration of Main Steam Isolation Valves, as a part of scram subsequent actions, a second reactor scram occurred due to low reactor water level.

The load shed is attributed to a defective light bulb being placed in the load shed panel. The shorted light bulb caused an overcurrent which subsequently caused degradation of a computer chip which initiated the load shed. The cards which contained degraded computer chips, due to the overcurrent, were replaced. The Division I Load Shed Panel was tested satisfactorily and operability was verified.

All safety systems functioned as designed. The minimum water level reached was -54.1 inches which was approximately 112 inches above the top of active fuel.

END OF ABSTRACT

TEXT PAGE 2 OF 4

A. Reportable Occurrence

On September 16, 1990 at 2019 hours, a reactor scram resulted from a Reactor Protection System (RPS) (EIIS Code: JC) actuation due to low Turbine Control (EHC) System (EIIS Code: TG) fluid pressure. Subsequent to the RPS actuation, a low reactor water level caused a High Pressure Core Spray (HPCS) System (EIIS Code: BG) actuation. During the scram recovery, another RPS actuation occurred due to low water level. These conditions are reported pursuant to 10CFR50.73(a)(2)(iv).

B. Initial Condition

The plant was operating in Mode I at approximately 93% power. A monthly surveillance was being performed on the Load Shedding and Sequencing (LSS) System (EIIS Code: EB).

C. Description of Occurrence

On September 16, 1990 during a monthly surveillance of the LSS System, operations personnel observed the Offsite Power Available (OPA) # 1 indication light extinguished on the Div. I LSS panel. The bulb was replaced with a new bulb, but the indicating light still did not illuminate. The Div. I LSS portion of the surveillance was terminated and a Condition Identification was initiated to document the condition. The non-licensed operator then proceeded to the Div. II LSS panel to perform the appropriate portion of the surveillance. While performing the Div. II LSS portion of the surveillance, the Div. I LSS System caused a Balance of Plant (BOP) load shed. The BOP load shed resulted in the loss of four BOP Busses (11HD, 12HE, 13AD, 14AE). Major plant components lost as a result of the load shed are as follows:

System/Components EIIS Code

Reactor Recirculation (Recirc) Pumps AD
Reactor Protection Motor/Generator Sets JC

Reactor Water Clean-Up Pump CE
Condensate and Condensate Booster Pumps SD
Heater Drain Pumps SN
Turbine Control Fluid Pumps TG

A rapid power reduction occurred following the loss of both Recirc pumps. Even though the Recirc pump trips caused a momentary swell in level, the level began to, decrease due to the Main Steam Line Isolation Valves (MSIVs) remaining in the open position, and the condensate and feedwater "systems being lost. The EHC fluid pressure decreased, as a result of the tripped EHC pumps, to the RPS scram setpoint which indicated a, fast closure of the Main Turbine Control Valves (TCVs). The reactor scramed on the TCVs fast closure as sensed from low EHC fluid pressure.

Following the closure of the TCVs, reactor pressure increased to the relief setpoint of Safety Relief Valve (SRV) 1B21F051D. The opening of this SRV initiated the low-low set relief logic. Ten additional valves opened when their relief setpoints were reached. The peak reactor pressure reached was 1119 psig.

The loss of the RPS M/G sets deenergized the MSIV solenoids which resulted in MSIV closures Due to the loss of inventory via the SRVs with no feedwater flow the Reactor Core Isolation Cooling (RCIC) System was manually initiated to maintain vessel level. Reactor water level continued to decrease until level 2 (-41.6 inches) was reached which caused an automatic actuation of HPCS. The minimum level reached during the occurrence was -54.1 inches. The cooler water being introduced to the vessel by HPCS and RCIC caused vessel pressure to decrease resulting in SRV closures. The inventory being loss through the SRVs was subsequently decreased and vessel level began to increase until the high level setpoint was reached. The HPCS injection valve was manually isolated and the RCIC steam supply valve closed as designed on high water level. The BOP busses were reenergized and power was successfully restored to the previously lost loads.

During the scram recovery, the scram was reset and the Division I Residual Heat Removal (RHR) System (EIIS Code: BO) was placed in the Suppression Pool Cooling mode of operation. SRV 1B21F051D was cycling to control vessel pressure.

As part of the scram recovery, an effort was made to open the MSIVs to establish reactor vessel pressure control. During pressure equalization across the MSIVs, reactor water level decreased to level 3 (+11.4 inches), the RPS low level scram setpoint, which resulted in a second reactor scram. RCIC was manually initiated and restored vessel level. Reactor water level increased until RCIC isolated on high water level. Subsequently, the Div. II RHR System was placed in the Suppression Pool Cooling mode of operation. The MSIVs were reopened and the Main Turbine Bypass Valves were used to control vessel pressure. The condensate and feedwater systems were restored to establish reactor level control.

D. Apparent Cause

After an extensive investigation, it was discovered that the operation of+ a faulty undervoltage relay, associated with the incoming breaker to bus 15AA (152-1501), caused the OPA # 1 circuit to be intermittently deenergized. Believing the OPA # 1 indicating light bulb was blown, the operator replaced this bulb with a new bulb which contained an internal short. When energized, the circuit experienced an overcurrent condition which failed gates on several computer chips. Subsequently, a gate failed which initiated the BOP load shed logic.

E. Supplemental Corrective Actions

The cards which contained degraded computer chips, due to the overcurrent caused by the shorted light bulb, were replaced. Bench testing was performed to determine the failed circuits on each replaced card. The Div. I LSS panel was tested satisfactorily and operability was verified.

The under voltage relay contacts, which caused the OPA # 1 indicator circuit to be deenergized, were cleaned and tested satisfactorily.

To enhance personnel awareness and to establish Administrative Controls over changing light bulbs on the LSS panel, a cautionary warning plaque has been placed on each panel. These plaques require Operations personnel to initiate a condition identification in order to replace a light bulb. This will in effect make this activity an orderly controlled evolution.

F. Safety Assessment

The Post Trip Analysis confirmed that the safety systems functioned properly. RCIC was manually initiated and operated as designed.

Additionally, HPCS automatically initiated on a low reactor water level of -41.6 inches and injected into the vessel. The HPCS diesel generator also automatically started due to the HPCS initiation. The reactor level remained at least 112 inches above the top of active fuel during the event. All Low Pressure Emergency Core Cooling Systems were operable but were not required to be automatically or manually initiated.

ATTACHMENT 1 TO 9010260079 PAGE 1 OF 1

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October 16, 1990 W. T. Cottle
Vice President

Operations Grand Gulf Nuclear Station

U.S. Nuclear Regulatory Commission
Mail Station P1-137
Washington, D.C. 20555

Attention: Document Control Desk

Gentlemen:

SUBJECT: Grand Gulf Nuclear Station
Unit 1
Docket No. 50-416
License No. NPF-29
Reactor Scram Due To
Loss Of BOP Busses
LER 90-017
AECM-90/0188

Attached is Licensee Event Report (LER) LER 90-017 which is a final report.

Yours truly,

WTC/RR:cg
Attachment

cc: Mr. D. C. Hintz (w/a)

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